company. Repair work done for that oil company since West State's bankruptcy has all been performed at US shipyards. Therefore, workers cannot be considered to have been adversely impacted by a shift in production to Canada or Mexico or by imports from those countries.

NAFTA–TAA–00475; Dante Fashions Corp., Jeannette, PA

The investigation revealed that criteria (3) and (4) were not met. Survey results revealed that customers do not import articles like or directly competitive with women's apparel from Canada or Mexico.

NAFTA–TAA–00482; Bill Neubert Log, Inc., Klamath Falls, OR

The investigation revealed that criteria (3) and (4) were not met. The investigation revealed that the customer to whom Bill Neubert Log, Inc. supplied contract logging did not import contract logging from Canada or Mexico.

NAFTA–TAA–00483; B & G Equipment Co., Plumsteadville, PA

The investigation revealed that criteria (3) and (4) were not met. A survey conducted with a major customer of consumer plastic sprayers revealed decreased purchases from the B&G Equipment Co. Also, this customer has not directly or indirectly purchased consumer plastic sprayers from Canada, Mexico or any other foreign source.

NAFTA–TAA–00487; Palliser Grain Co., Ltd., United States Office Great Falls, MT

The investigation revealed that the workers of Palliser Grain Co., Ltd, Great Falls, MT do not produce an article within the meaning of Section 2509(a) of the Trade Act, as amended.

### Affirmative Determination NAFTA–TAA

NAFTA–TAA–00473; Cowlitz Stud Co., Randle & Morton Div., Randle, WA

A certification was issued covering all workers at Randle & Morton Divisions of Cowlitz Stud Co, Randle and Morton, WA separated on or after May 24, 1994.

NAFTA–TAA–00470; Seagull Energy Corp./Midcon, Inc., Amarillo, TX

A certification was issued covering all workers at Seagull Energy Corp./ Midcon, Inc., Amarillo, TX separated on or after May 15, 1994.

NAFTA–TAA–00469; Planergy New York, Inc., East Syracuse, NY

A certification was issued covering all workers at Planergy New York, Inc., East Syracuse, NY separated on or after May 23, 1994.

NAFTA–TAA–00467; Vernitron Corp., St. Petersburg, FL

A certification was issued covering all workers at VERNITRON/VRN International, St. Petersburg, FL separated on or after May 22, 1994.

NAFTA–TAA–00500; Occidental Chemical Corp., Durez Div., North Tonawanda, NY

A certification was issued covering all workers engaged in the production of phenolic molding compounds at the Occidental Chemical Corp., North Tonawanda, NY separated on or after May 30, 1994.

NAFTA–TAA–00494 & A; Miniature Precision Components, Inc., Walworth, WI & Prairie De Chien, WI

A certification was issued covering all workers at the Miniature Precision Components, Inc., Walworth and Prairie De Chien, WI separated on or after June 20, 1994.

NAFTA–TAA–00477 & A, B, C; Crown Pacific Limited Partnership, Colburn Unit, Sandpoint, ID, Bonners Ferry ID, Thompson Falls, MT, & Operating in the States of ID, MT and WA

A certification was issued covering all workers of Crown Pacific Limited Partnership, Sandpoint and Bonners Ferry, ID & Thompson Falls, MT and other locations operating in ID, MT and WA separated on or after May 25, 1994.

NAFTA–TAA–00504; Nashua Corp., Nashua Cartridge Products, Inc., Exeter, NH

A certification was issued covering all workers of Nashua Cartridge Products, Inc., of the Nashua Corp, Exeter, NH separated on or after June 23, 1994.

NAFTA–TAA–00486 A, B; Equitable Resources Energy Co., Equitable Resources Exploration Div., Kingsport, TN, Nora VA and Hazard, KY

A certification was issued covering all workers at the Equitable Resources Exploration Div. of the Equitable Resources Energy Co., Kingsport, TN, Hazard, KY and Nora, VA separated on or after June 12, 1994.

NAFTA–TAA–00498; Takata, Inc, Gateway Safety Systems, Michigan City, IN

A certification was issued covering all workers at the Gateway Safety Systems division of Takata, Inc., Michigan City, IN separated on or after June 15, 1994.

I hereby certify that the aforementioned determinations were issued during the month of July, 1995. Copies of these determinations are available for inspection in Room C–4318, U.S. Department of Labor, 200 Constitution Avenue, NW., Washington, DC 20210 during normal business hours or will be mailed to persons who write to the above address.

Dated: July 12, 1995.

**Victor J. Trunzo,**

*Program Manager, Policy & Reemployment Services, Office of Trade, Adjustment Assistance.*

[FR Doc. 95–17734 Filed 7–18–95; 8:45 am]

BILLING CODE 4510–30–M

## NUCLEAR REGULATORY COMMISSION

### Biweekly Notice

### Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

### I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 23, 1995, through July 7, 1995. The last biweekly notice was published on July 5, 1995 (60 FR 35058).

### Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By August 18, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

**Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois**

*Date of application for amendment request:* September 10, 1993, as supplemented June 16, 1995.

*Description of amendment request:* As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The September 10, 1993, and June 16, 1995, applications proposed to upgrade only Section 3/4.8 (Plant Systems) of the Dresden and Quad Cities TS.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident. Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.8 are based on STS guidelines or later operating BWR plant's NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Plant Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and, therefore, reduce the consequences of any accident previously evaluated, as the probability of the systems outlined within Section 3/4.8 of the proposed Technical Specifications, performing their intended function is increased by the additional surveillances.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.8 is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden or Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations.

No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Plant Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Plant Systems are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.8 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden or Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden or Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Plant Systems when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

*Attorney for licensee:* Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

*NRC Project Director:* Robert A. Capra

**Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois**

*Date of application for amendment requests:* September 17, 1993, as supplemented June 30, 1995

*Description of amendment requests:* As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operation and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The September 17, 1993, and June 30, 1995, applications proposed to upgrade only Section 3/4.6 (Primary System Boundary) of the Dresden and Quad Cities TS.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendments for Dresden and Quad Cities Station's Technical Specification Section 3/4.6 are based on STS guidelines or later operating BWR plant's NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Primary System Boundary are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and therefore, reduce the consequences of any accident previously evaluated as the probability of the systems outlined within Section 3/4.6 of the proposed Technical Specifications, performing its intended function is increased by the additional surveillances.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.6 is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Primary System Boundary are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Primary System Boundary are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements.

However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.6 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Primary System Boundary when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

*Local Public Document Room location:* for Dresden, Morris Public Library, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

*Attorney for licensee:* Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

*NRC Project Director:* Robert A. Capra

**Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas**

*Date of amendment request:* March 17, 1995

*Description of amendment request:* The proposed amendment transfers requirements for a cycle specific core operating limit from the Technical Specifications to the Core Operating Limits Report. Additionally, a reference to a statistical methodology for determining uncertainties is being changed to reference a methodology that was recently approved by the NRC.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability of an Accident Previously Evaluated.

The removal of the cycle-dependent value for the departure from nucleate boiling ratio (DNBR) reduction from technical specifications and placing it into the Core Operating Limits Report (COLR) has no impact on plant operation or accident analyses. The proposed change is considered to be administrative in nature. Technical specifications will continue to require operation within the core operational limits for each cycle reload calculated by the approved reload design methodologies. The appropriate actions required if limits are violated will remain in the technical specifications. The reload report presents the results of a cycle-specific evaluation of accidents and transients addressed in the ANO-2 Safety Analysis Report (SAR). The cycle-specific evaluation demonstrates that changes in the fuel cycle design and the corresponding COLR do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Modified Statistical Combination of Uncertainties (MSCU) methodology statistically combines uncertainties to at least a 95/95 probability/confidence level. The Proposed change to reference the MSCU is administrative in nature. The currently referenced methodology is being replaced with a more recently approved methodology which has been determined to be applicable to ANO-2. The new methodology has been independently reviewed and approved by the NRC. This change does not impact either the manner in which the operating margin to limits on linear heat rate and DNBR is maintained or the manner in which the CPCs respond to transients and provide trips. Therefore, this change does not adversely impact transient analysis assumptions or results. In addition, the physical design or operation of the plant is not impacted by this change. The safety analyses will continue to be performed utilizing NRC-approved methodologies and specific reload changes will be evaluated per 10CFR50.59.

Therefore, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change to relocate the cycle-specific value for the DNBR reduction from technical specifications to the COLR is administrative in nature. No change to the design, configuration, or method of operation of the plant is made by this change. This parameter will be determined using NRC-approved methods. Technical specifications will continue to require operation within the required core operating limits and appropriate actions will be taken if the limits are exceeded. The relocation of a cycle-specific parameter does not create the possibility of a new or different of accident from any accident previously evaluated.

The proposed change to reference the NRC-approved MSCU methodology is administrative in nature. The currently referenced methodology is being replaced with a more recently approved methodology which has been determined to be applicable to ANO-2. No physical alterations of plant configuration, changes to plant operating procedures, or operating parameters are proposed. The safety analyses are still performing utilizing NRC-approved methodologies and specific reload changes will be evaluated per 10CFR50.59. No new equipment is being introduced, and no equipment is being operated in a manner inconsistent with its design.

Therefore, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

Existing technical specification operability and surveillance requirements are not reduced by the proposed change to relocate the cycle-specific value for DNBR reduction to the COLR. The development of limits for a particular cycle will continue to conform to methods described in NRC-approved documentation. Technical specifications will still require that the core be operated within these limits and specify appropriate actions to be taken if the limits are violated. The cycle-specific COLR limits for future reloads will be developed based on NRC-approved methodologies. Each reload undergoes a 10CFR50.59 safety review to assure that operating of the unit within the cycle-specific limits will not involve a significant reduction in a margin of safety.

The proposed change to reference the MSCU methodology is administrative in nature. The currently referenced methodology is being replaced with a more recently approved methodology which has been determined to be applicable to ANO-2. The resultant overall uncertainty factors using the MSCU methodology are determined and applied to at least the same 95/95 probability/confidence level as the overall uncertainty factors using the current methodology. NRC review and approval of the methodologies used to perform the cycle-specific reload analyses is not affected by this change. The safety analyses are still performed utilizing NRC-approved methodologies and specific reload changes will be evaluated per 10CFR50.59.

Therefore, these changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

*NRC Project Director:* William D. Beckner

**Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas**

*Date of amendment request:* March 17, 1995

*Description of amendment request:* The proposed amendment deletes requirements associated with surveillances to verify position stops for High Pressure Safety Injection Emergency Core Cooling System throttle valves.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The HPSI system is not an initiator of a previously evaluated accident; therefore, the probability of a previously evaluated accident will not be increased by the proposed change. Accidents which require the use of HPSI will not have any increased consequences since the new injection/isolation valve arrangement is at least as reliable as the previous valve arrangement. No part of the proposed change has any adverse effect upon the HPSI system response or function. The new manual valves will perform the throttling function previously performed by the HPSI isolation MOVs without reliance upon any electrical equipment (MOV limit switches). The proposed change does not affect routing of HPSI piping or affect total flow characteristics of the system. The proposed change to remove the requirement to verify the correct settings of position stops for the HPSI throttle valves is consistent with NUREG-1432, restructured ''Standard Technical Specifications - Combustion Engineering Plants,'' since the manual throttle valves fixed into position serve the function of, and are equivalent to, flow limiting orifices.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change does not change the function or mode of operation of the HPSI system. The failure of the new MOVs to function will have no different effect than failure of the previously installed MOVs and such failure is enveloped by assumptions in the existing safety analysis, i.e., redundant trains will still be able to function. The new manual valves are less likely to fail in operation since they are fixed into position by tack-welded locking devices and therefore perform their function passively. Inadvertent manipulation of the manual valves will be prevented by the locking arrangement. There are no new functions or modes being accomplished by the MOVs. The throttling function to be performed by the manual valves will be more reliably performed by passive components than by active electrical circuits. The change eliminates uncertainty in throttle valve position as a result of limit switch tolerances and repeatability which form the basis for the current surveillance requirement for periodic verification.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.No margin of safety will be reduced or affected by the proposed deletion of the surveillance requirement. The new manual valves will be throttled to produce a system flow balance equivalent to the current one, and the balance will continue to be confirmed by surveillance testing in accordance with TS requirements.

Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

*NRC Project Director:* William D. Beckner

**Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas**

*Date of amendment request:* March 17, 1995

*Description of amendment request:* The proposed amendment revises requirements associated with the frequency of containment post-entry visual inspections.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change to the Arkansas Nuclear One-Unit 2 (ANO-2) technical specifications (TS) does not involve any system or component or condition evaluated as an accident initiator; therefore there is no increase in the probability of an accident previously evaluated.

The purpose of this change is to reduce the required number of containment inspections following entries at operational modes above cold shutdown. This reduction in the number of inspections will reduce personnel exposure to radiation and potential heat stress. These inspections are to verify that no debris that might be transported to the containment sump is left behind at the conclusion of the entry. Typically, containment entries above cold shutdown are for specific purposes and involve a limited area of containment. The expectation for job performance at ANO-2 is that a job site is left cleaner than found. The inspection serves as a verification that any materials taken into the containment building which might foul the sump screens have been removed or have been properly anchored. Performing this inspection on a daily frequency will not result in changing the work practices at ANO-2, therefore the amount of debris generated or left in containment should not increase. The daily inspection will be sufficient verification that conditions in containment are not degrading; therefore, there will be no significant increase in the consequences of an accident previously evaluated.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

Because the proposed amendment will not change the design, configuration, or method of operation of the plant, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

There will be no adverse effects on margins of safety since materials that are considered acceptable to remain in containment has not changed. By reducing the number of inspections, no mechanism has been created that will generate more debris in containment nor have work practices been altered to allow less stringent controls over what is taken in or left in containment. Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

*NRC Project Director:* William D. Beckner

**Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas**

*Date of amendment request:* April 4, 1995

*Description of amendment request:* The proposed amendment deletes requirements associated with part length control element assemblies. During the upcoming refueling outage all part length control assemblies will be removed from the reactor.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed changes maintain conservative restrictions on the operation of those control element assemblies (CEAs) formerly specified as part length CEAs (PLCEAs) and are considered to be administrative in nature. The Arkansas Nuclear One - Unit 2 (ANO-2) Safety Analysis Report (SAR) Chapter 15 accident analyses identify four families of analyses associated with the CEAs. Each of these analyses is evaluated in the development of the Reload Report for each fuel cycle, and the appropriate limitations to insure acceptable analysis results are incorporated in the Core Operating Limits Report (COLR) for the fuel cycle. The modification replacing the PLCEAs with full length CEAs will be evaluated under the Arkansas Nuclear One (ANO) 10CFR50.59 process prior to implementation. The Reload Report and changes to the COLR are also evaluated under the ANO 10CFR50.59 process prior to incorporating the identified changes. Movement of the PLCEAs during power operation has typically resulted in more dropped CEAs than movement of the full length CEAs due to the greater weight of the PLCEAs. Replacement of the PLCEAs with full length CEAs should result in a reduction in the probability of a dropped CEA.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes introduce no new mode of plant operation and are considered to be administrative in nature. Operating experience has shown that the full length CEAs are capable of controlling the axial power distribution function intended for the PLCEAs. The PLCEAs will be replaced with the same type of full length CEAs used in shutdown and regulating CEA groups.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin Safety.

The proposed changes may improve overall safety margins. Replacement of the PLCEAs with full length CEAs and including these Group P CEAs in the CEA drop time testing will allow ANO-2 to credit these CEAs in the shutdown margin calculations. This should result in an increase in the available shutdown margin during reactor operation.

Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

*NRC Project Director:* William D. Beckner

**Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas**

*Date of amendment request:* April 4, 1995

*Description of amendment request:* The proposed amendment revises the containment cooling response time to reduce the likelihood of a water hammer event in service water piping.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or consequences of an Accident Previously Evaluated.

The containment cooling system and the service water system are not considered to be accident initiators for any analyzed accident. The containment cooling system functions to mitigate the effects of a Main Steam Line Break (MSLB) or Loss of Coolant Accident (LOCA) on the containment environment. The proposed change does not affect the limiting MSLB analysis as the proposed increase in containment cooling response time is only instituted on a loss of off-site power. The limiting LOCA analysis has been evaluated with respect to the proposed containment cooling response time. Although the analysis shows an increase in the containment peak pressure (approximately 0.1 psig), this increase in the peak containment pressure is not considered significant since the MSLB accident with off-site power available is still the overall limiting accident condition with respect to containment peak pressure. The containment peak conditions for the LOCA and MSLB analyses remain below the original Final Safety Analysis Report (FSAR) conditions of 53.4 psig and 288°F.

Therefore, this change does not involve a significant increase int he probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change in containment cooling response time introduces no new mode of plant operation. Containment cooling response time is an analytical input and is not considered to be the initiator of any accident condition.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The increase in containment cooling response time has been evaluated with respect to the accident analyses resulting in peak containment pressures. This evaluation has shown no significant increase in the resulting peak containment pressure since the overall limiting accident with respect to containment pressure is still the MSLB with off-site power available. The containment peak conditions for the LOCA and MSLB analyses remain below the original FSAR conditions of 53.4 psig and 288°F.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

*NRC Project Director:* William D. Beckner

**Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana**

*Date of amendment request:* May 25, 1995

*Description of amendment request:* The proposed amendment revises the Physical Security Plan vital island requirements.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The accident mitigation features of the plant are not affected by the proposed change. This change provides an equivalent level of protection to the plant and is adequate for preventing an unacceptable risk to public health and safety. This is due to continued compliance with existing regulatory requirements, the integral defense in depth design of the security program, including programs in place to minimize the threat of insiders, and historically high system reliability. The SBO (Station Blackout diesel) is designed with sufficient capacity to accommodate station blackout needs as well as those required for security. Ample protection against a design basis security threat continues to be provided. Therefore, this change does not increase the probability or consequences of an accident previously evaluated.

The Station Blackout diesel generator has been approved and accepted by the Staff pursuant 10CFR50.63. New systems, modes of equipment operation, failure modes, or other plant perturbations are not introduced by this change. The change provides an equivalent level of protection, does not decrease the effectiveness of the overall security program and is adequate for preventing an unacceptable risk to public health and safety. Ample protection against a design basis security threat continues to be provided with overall physical protection of the plant maintained. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

This change does not change a safety limit, an LCO (Limiting Condition of Operation), or a surveillance requirement on equipment required to operate the plant. It is equivalent in level of protection, does not decrease the effectiveness of the security program and is adequate for preventing an unacceptable risk to public health and safety. The SBO diesel generator will provide an adequate alternative source of power to security systems. Ample protection against a design basis security threat continues to be provided. Therefore, this change does not involve a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803

*Attorney for licensee:* Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

*NRC Project Director:* William D. Beckner

**Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas**

*Date of amendment request:* May 22, 1995

*Description of amendment request:* The proposed amendment would revise Technical Specification (TS) 4.8.1.1.2.e.7 to allow the performance of the 24-hour surveillance test of the diesel generators during power operation.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change to permit the 24 hour surveillance test of the diesels to be performed during power operation does not increase the chances for a previously analyzed accident to occur. The function of the diesels is to supply emergency power in the event of a loss of offsite power. Operation of the diesels is not a precursor to any accident. Furthermore, the diesel generator being tested will remain operable and will be available to supply emergency loads within the required time. In addition, the two remaining diesel generators will be operable during the test. Consequently, if an offsite disturbance were to occur that affected the operability of the diesel being tested, the two remaining diesels would be capable of feeding the loads necessary for safe shutdown of the plant. This addresses the concerns raised in Information Notice 84-69 regarding the operation of emergency diesel generators connected in parallel with offsite power. In summary, the proposed changes do not adversely affect the performance or the ability of the diesel generators to perform their intended function.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment to the 24 hour surveillance test will not affect the operation of any safety system or alter its response to any previously analyzed accident. The diesel will automatically transfer from the test mode if necessary to supply emergency loads in the required time. The test mode is used for the monthly surveillance of the diesel generators as well, therefore, no new plant operating modes are introduced. In the event the diesel fails the surveillance test, it will be declared inoperable and the actions required for an inoperable diesel will be performed. The remaining two diesel generators will be operable and are capable of feeding the loads necessary for safe shutdown of the plant.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

The proposed amendment will not reduce availability of the diesel generator being tested to provide emergency power in the event of a loss of offsite power. If a loss of offsite power occurs during the surveillance test, the diesel generator output breaker will be tripped by the directional over-current relay on the ESF transformer. The diesel generator will transfer to the emergency mode, and the ESF undervoltage logic will initiate Mode II (Loss of Offsite Power) operation of the ESF load sequencer to supply emergency loads from the diesel generator. If a Loss of Coolant Accident occurs during the surveillance test, the diesel generator output breaker will be opened by a signal from the Solid State Protection System and the preferred offsite source will continue to provide power to the ESF bus. The diesel generator will continue to run in the emergency mode and would be available to automatically supply safety-related loads during any loss of offsite power condition. The test mode to emergency mode transfer is tested once per cycle in accordance with Surveillance Requirement 4.8.1.1.2.e.10. In addition, the two remaining generators will be operable during the test. Consequently, if an offsite disturbance were to occur that affected the operability of the diesel being tested, the two remaining diesels would be capable of feeding the loads necessary for safe shutdown of the plant. The time required for the diesel being tested to pick up emergency loads will not be affected by performing the 24 hour surveillance test during power operation.

Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

*Local Public Document Room location:* Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

*Attorney for licensee:* Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

*NRC Project Director:* William D. Beckner

**Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas**

*Date of amendment request:* May 25, 1995

*Description of amendment request:* The proposed amendment would revise the technical specifications (TSs) on containment leakage, to make the action statement consistent with the need to perform Type C testing at power, and to replace the surveillance requirements with a single requirement to apply the requirements of Appendix J as modified by approved exemptions. The proposed amendment would also revise the TSs on containment integrity, containment leakage, and containment air locks, to eliminate the numerical value of calculated peak containment internal pressure related to the design basis accident. In addition, there is an associated proposed exemption, from the requirements of 10 CFR Part 50, Appendix J, to allow the performance of the required periodic Type C tests during power operation.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change to the action statement of Technical Specification 3.6.1.2 does not significantly increase the probability of an accident because leakage rate testing is not an accident initiator. The consequences of an accident previously evaluated are not increased by changing the ACTION statement of Technical Specification 3.6.1.2 because the requirements for CONTAINMENT INTEGRITY are not reduced. The consequences of an accident previously evaluated are not increased by the change in the surveillance wording because no technical changes are proposed. The underlying purpose of the proposed change to the Technical Specifications and requested exemption to Appendix J, to allow surveillance credit for at-power Type C testing, will not increase the consequences of an accident because there are no reductions in the requirements to maintain containment integrity.

The proposed change to delete the numeric value of Pa is purely administrative, and has no potential effect on accident initiation or consequences.

2. Does the change create the possibility of a new or different kind of accident from any previously evaluated?

Nothing associated with the requested changes will physically change the configuration of the plant or impose new operating configurations not previously considered. Leakage rate testing will remove components and trains from service; however, this is not operationally different from other testing and maintenance evolutions that remove components or trains from service, and which were previously considered. Consequently, the possibility of a new or different kind of accident from any previously evaluated is not created.

3. Does this change involve a significant reduction in the margin of safety?

The margin of safety is not significantly reduced by changing the ACTION statement of Technical Specification 3.6.1.2 because the requirements for CONTAINMENT INTEGRITY are not reduced. The margin of safety is not reduced by the change in the surveillance wording because no technical changes are proposed. The underlying purpose of the proposed change to the Technical Specifications and requested exemption to Appendix J, to allow surveillance credit for at-power Type C testing, will not reduce the margin of safety because there are no reductions in the requirements to maintain containment integrity.

The proposed change to delete the numeric value of $P_a$ is purely administrative, and has no potential effect on the margin of safety because the value itself is unchanged.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

*Local Public Document Room location:* Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

*Attorney for licensee:* Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

*NRC Project Director:* William D. Beckner

**Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas**

*Date of amendment request:* May 30, 1995

*Description of amendment request:* The proposed amendment would increase the spent fuel pool heat load licensing basis to provide greater flexibility for normal refueling practices.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

(a) The Spent Fuel Pool conditions are not indicative of accident initiators.

(b) Design and operability requirements of equipment important to safety are not affected.

(c) If only one Spent Fuel Pool cooling train is available, boiling would not occur and the Spent Fuel Pool components would remain within their design basis.

(d) The complete loss of Spent Fuel Pool cooling event has previously been analyzed and described in Supplement 6 to the Safety Evaluation Report, Appendix BB. The dose consequences for this event have been evaluated and the safety evaluation is described in Updated Safety Analysis Report Section 9.1.3.3.4. The results of the evaluation show that the Spent Fuel Pool components would remain within their design bases. Also, the dose consequences of iodine release as a result of Spent Fuel Pool boiling are significantly below the allowable dose limits of 10 CFR 100.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously because:

(a) The operability of safety-related equipment is not impacted.

(b) The probability of safety-related equipment malfunctioning is not increased.

(c) The scope of the change does not establish a potential new accident precursor.

(d) The Spent Fuel Pool design considers design basis heat loads for the modified refueling procedure which includes a full-core offload.

(e) For the design basis case, the integrity of the Spent Fuel Pool Boraflex is not adversely impacted.

3. The proposed changes do not involve a significant reduction in the margin of safety because:

(a) No fuel damage would occur as a result of the proposed change.

(b) Technical Specification operability and surveillance requirements are not reduced.

(c) The Spent Fuel Pool boiling doses would be significantly below the allowable dose limits of 10 CFR 100.

(d) The modified refueling procedure (full-core offload) continues to have acceptable margins of safety.

(e) For the design basis case, the integrity of the Spent Fuel Pool Boraflex is not adversely impacted.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

*Local Public Document Room location:* Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

*Attorney for licensee:* Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

*NRC Project Director:* William D. Beckner

**Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois**

*Date of amendment request:* June 9, 1995

*Description of amendment request:* The proposed amendment would modify Technical Specification (TS) 4.1, ''Site Location,'' to incorporate a description of the exclusion area boundary. The proposed change is necessary to ensure the content of the TS conforms to Section 182 of the Atomic Energy Act of 1954.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

(1) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents previously evaluated. In addition, the physical location of the [exclusion area boundary] EAB has not been changed; a description of its location has merely been added to the TS. Thus, the proposed change cannot increase the probability or the consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameter or condition that could contribute to the initiation of any accidents. Thus, the proposed change cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed change only affects regulatory controls on the accepted configuration of the EAB. The proposed change does not involve an actual change to the location of the EAB. The proposed change will restore compliance with the Atomic Energy Act of 1954 and require prior NRC approval of any changes to the physical location of the EAB. As a result, IP has concluded that the proposed change will not result in a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

*Attorney for licensee:* Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th St., Decatur, Illinois 62525

*NRC Project Director:* Gail H. Marcus

**Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit No. 1, Berrien County, Michigan**

*Date of amendment request:* February 3, 1995, as supplemented April 25, 1995 (AEP:NRC:1166Q and 1166R)

*Description of amendment request:* The proposed amendment would allow continued use of a steam generator (SG) tube support plate interim plugging criteria for fuel cycle 15. The change would allow SG tubes with bobbin coil eddy current indications less than or equal to 2.0 volts at tube support plate intersections to remain in service if the projected end-of-cycle distribution of crack indications is shown to result in primary-to-secondary leakage less than 12.6 gpm during a postulated steam line break (SLB). The change would also allow indications greater than 2.0 volts but less than or equal to 5.6 volts to remain in service if a motorized rotating pancake coil probe inspection does not detect degradation.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in margin of safety. Conformance of the proposed amendment to the standards for a determination of no significant hazards as defined in 10 CFR 50.92 (three factor test) is shown in the following paragraphs.

1) Operation of Cook Nuclear Plant Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Testing of model boiler specimens for free span tubing (no tube support plate restraint) at room temperature conditions show burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on pulled tubes from Cook Nuclear Plant Unit 1 with up to a 2.02 volt indication shows measured burst pressure in excess of 10,000 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show burst pressures in excess of 6,300 psi at room temperatures. Correcting for the effects of temperature on

material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety factor requirements of RG [Regulatory Guide] 1.121 [''Bases for Plugging Degraded PWR Steam Generator Tubes'']. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate [TSP]. Test data indicates that tube burst cannot occur within the tube support plate, even for tubes which have 100% throughwall electric-discharge machined notches 0.75 inch long, provided the tube support plate is adjacent to the notched area. Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must, therefore, retain tube integrity characteristics which maintain the RG 1.121 margin of safety of 1.43 times the bounding faulted condition (steam line break) pressure differential.

During a postulated main steam line break, the TSP has the potential to deflect during blowdown, thereby uncovering the intersection. Based on the existing data base, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the steam line break pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than 9.6 volts, regardless of the indicated depth measurement. A 2.0 volt plugging criteria compares favorably with the 9.6 volt structural limit considering the previously calculated growth rates for ODSCC [outer diameter stress corrosion cracking] within Cook Nuclear Plant Unit 1 SGs. Considering a voltage growth component of 0.8 volts (40% voltage growth based on 2.0 volts BOC [beginning of cycle]), and a nondestructive examination uncertainty of 0.40 volts (20% voltage uncertainty based on 2.0 volts BOC), when added to the BOC IPC [interim plugging criteria] of 2.0 volts, results in a bounding EOC [end of cycle] voltage of approximately 3.2 volts for cycle 15 operation. A 6.4 volt safety margin exists (9.6 structural limit - 3.2 volt EOC - 6.4 volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 9.6 volts. Using this structural limit of 9.6 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit a significant number of EOC indications to exceed the 9.6 volt structural limit and should assure that acceptable tube burst probabilities are attained. By adding NDE [nondestructive examination] uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. The previous IPC submittal established the conservative NDE uncertainty limit of 20% of the BOC repair limit. For consistency, a 40% voltage growth is extremely conservative for Cook Nuclear Plant Unit 1. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 9.6 volts can be represented by the expression:

RL + (0.2 x RL) + (0.4 x RL) = 9.6 volts, or,the maximum allowable BOC repair limit can be expressed as,

RL = 9.6 volt structural limit/1.6 = 6.0 volts

This structural repair limit supports this application for cycle 15 IPC implementation to repair bobbin indications greater than 2.0 volts based on RPC [rotating pancake coil] confirmation of the indication. Conservatively, an upper limit of 5.6 volts will be used to repair bobbin indications which are above 2.0 volts but do not have confirming RPC calls.

The conservatism of this repair limit is shown by the EOC 13 (Spring 1994) eddy current data. The overall average voltage growth was determined to be on 1.4% (of the BOC voltage). In addition, the EOC 13 maximum observed voltage increase was 0.40 volts, and occurred in a tube with a BOC indication of 0.96 volts. The applicability of cycle 14 growth rates for cycle 15 operation will be confirmed prior to return to service of Cook Nuclear Plant Unit 1. Similar large structural margins are anticipated.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main steam line break outside of containment but upstream of the main steam isolation valve represent the most limiting radiological condition relative to the IPC. In support of implementation of the IPC, it will be determined whether the distribution of crack indications at the tube support plate intersections at the end of cycle 15 are projected to be such that primary to secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate calculation has determined this allowable steam line break leakage limit to be 12.6 gpm. Although not required by the Cook Nuclear Plant design basis, this calculation uses the recommended Iodine-131 transient spiking values consistent with NUREG-0800 [Standard Review Plan], and the T/S [technical specification] reactor coolant system activity limit of 1.0 micro curie per gram dose equivalent Iodine-131. The projected steam line break leakage rate calculation methodology prescribed in [Draft] GL 94-XX [''Voltage-Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' August 12, 1994] and WCAP 14277 [''SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections''] will be used to calculate EOC 15 leakage, based on actual EOC 14 distributions and EOC 15 projected distributions. Due to the relatively low voltage growth rates at Cook Nuclear Plant Unit 1 and the relatively small number of indications affected by the IPC, steam line break leakage prediction per GL 94-XX is expected to be significantly less than the acceptance limit of 12.6 gpm in the faulted loop.

Prior to issue of GL 94-XX, projected EOC 14 leak rates were calculated, based on draft NUREG-1477 [''Voltage-Based Interim Plugging Criteria for DG Tubes, Draft for Comments''], for a total of twelve cases, the combination of six probability-of-leak correlations and two leak rate calculation methodologies. Results of the calculations show that the projected EOC 14 leak rates ranged from 0.001 gpm to 1.360 gpm. These results are well below the 12.6 gpm allowable; therefore, implementation of the 2 volt IPC during cycle 15 would not adversely affect SG tube integrity and results in acceptable dose consequences.

Current GL 94-XX methodology requires only the log-logistic probability of leakage correlation be used. Projected EOC 14 SLB leakage using this function was calculated to be only 0.001 gpm. Based on the relatively few numbers of intersections at Cook Nuclear Plant Unit 1 to which the IPC are applied and extremely small Cook Nuclear Plant Unit 1 plant-specific growth rate, a similar value would be expected based on the EOC 14 eddy current data. The inclusion of all IPC intersections in the leakage model, along with application of a probability of detection of 0.6, will result in extremely conservative leakage estimations, especially so since close examination of the available data shows that indications of less than 2.8 volts will not be expected to leak during SLB conditions. All Unit 1 IPC indications are expected to be below 2.8 volts at the EOC 15 conditions.

The proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the Cook Nuclear Plant Unit 1 FSAR.

2) The proposed license amendment does not create the possibility of a new or different kind of accident previously evaluated.

Implementation of the proposed SG tube IPC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a SG in which the plugging criteria has been applied (during all plant conditions).

Specifically, we will continue to implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per SG to help preclude the potential for excessive leakage during all plant conditions. The cycle 15 T/S limits imposed on primary to secondary leakage at operating conditions are: a maximum of 0.4 gpm (600 gpd) for all SGs with a maximum of 150 gpd allowed for any one SG.

The RG 1.121 criteria for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. Regulatory Guide 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 9.6 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% prediction limit on the burst correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 9.6 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the ''longest permissible crack'' for evaluating operating leakage limits.

Consistent with the Cycle 13 and Cycle 14 license amendment requests for IPC and Section 5 of Enclosure 1 of the GL, operational leakage limits will remain at 150 gpd per SG. Axial cracks leaking at this level are expected to provide leak before break (LBB) protection at both the SLB pressure differential of 2560 psi and, while not part of any established LBB methodology, LBB protection will also be provided at a value of 1.43 times the SLB pressure differential. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steam line break conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection.

3) The proposed license amendment does not involve a significant reduction in margin of safety.

The use of the voltage based bobbin probe interim tube support plate elevation plugging criteria at Cook Nuclear Plant Unit 1 is demonstrated to maintain SG tube integrity commensurate with the criteria of RG 1.121. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting GDC [General Design Criteria] 14, 15, 31, and 32 by reducing the probability or the consequences of SG tube rupture. This is accomplished by determining the limiting conditions of degradation of SG tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. The EOC 15 distribution of crack indications at the tube support plate elevations will be confirmed by analysis and calculation to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of a LOCA [loss-of-coolant accident] and SSE [safe-shutdown earthquake] on the SG component (as required by GDC 2), it has been determined that tube collapse may occur in the SGs at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the Cook Nuclear Plant Unit 1 reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. Loss of coolant accident loads for the primary pipe breaks were used to bound the Cook Nuclear Plant Unit 1 smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in SG tube collapse or significant deformation.

Addressing RG 1.83 [''Inservice Inspection of PWR Steam Generator Tubes''] considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volts is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plant elevations per T/S, and MRPC [Motorized Rotating Pancake Coil] inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of SG tube plugs reduces the RCS flow margin. Thus, implementation of the IPC will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any Bases of the plant T/Ss.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

*Attorney for licensee:* Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW, Washington, DC 20037

*NRC Project Director:* John N. Hannon

**Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan**

*Date of amendment requests:* May 26, 1995 (AEP:NRC:1207)

*Description of amendment requests:* The proposed amendments would change multiple operating limits on both units. The primary change would allow operation of Cook Unit 1 with steam generator plugging levels as high as 30% in each steam generator. The second group of changes would modify the overtemperature delta T and overpower delta T reactor trip setpoints for both units and increase the allowed degradation of the Unit 1 auxiliary feedwater pumps consistent with Unit 2. The third group of changes would reduce the required shutdown margin in modes 1, 2, 3, and 4, increase the allowable centrifugal charging pump head degradation, reduce the minimum refueling water storage tank temperature, and revise the peak pressure of the long-term containment integrity analysis in the bases. Finally, certain administrative changes are also proposed.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

10 CFR 50.92 specifies that the holder of an operating license or construction permit of a nuclear power facility participate in determining whether a change to the T/S's current licensing basis (CLB) involves a significant hazards consideration. Prior to implementation of a change to the CLB, the Nuclear Regulatory Commission must review and make a final determination, pursuant to the procedures in 10 CFR 50.91, that a proposed amendment to the operating license involves no significant hazards considerations. In order to satisfactorily complete the review, the proposed amendment to the CLB must not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated,

2. create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. involve a significant reduction in a margin of safety.

For the purpose of performing a significant hazards consideration analysis, the four groups of technical specification changes discussed under Description of Changes can be reduced to three groups. In evaluating significant hazards, the first three groups of proposed technical specifications will be considered together. The miscellaneous change and the administrative change will each be considered separately.

*Determination Of No Significant Hazards For Changes Based On Analyses And Evaluations (Groups 1, 2, and 3)*

*Criterion 1*

Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The analyses which were performed to support the first three groups of proposed changes were performed in accordance with approved methodologies and acceptance criteria applicable to Cook Nuclear Plant. The proposed technical specification changes do not involve postulated initiators for analyzed events. Therefore, the probability of accidents can not be affected. The analyses and evaluations performed all met applicable acceptance criteria. Therefore, the consequences of accidents previously evaluated are unaffected.

Criterion 2

Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The analyses which were performed to support the second and third groups of proposed changes address increases in operating margin for accident mitigators. They do not create the possibility of new accidents. The first group of proposed changes to reduce minimum measured primary flow, increase the DNB [departure from nucleate boiling] temperature limit, and reduce the reactor coolant system volume have been analyzed or evaluated. The proposed DNB limit is consistent with the DNB design and does not constitute an accident initiator. The new volume results from the new value of allowed tube plugging and is consistent with the analysis. It is not an accident initiator.

The impact of the reduced primary flow in the primary system was analyzed or evaluated, as appropriate. All applicable criteria were satisfied. No new or different kind of accident resulted.

*Criterion 3*

Do the proposed changes involve a significant reduction in a margin of safety?

No. The margin of safety is provided for the primary pressure boundary and other components in part by applicable design codes. The margin of safety for the various accidents and transients is maintained by the analysis acceptance criteria. Since the components remain in compliance with the codes and standards in effect when Cook Nuclear Plant was licensed and applicable acceptance criteria are met, the margin of safety is not reduced by the 30% SGTP [steam generator tube plugging] program.

*Determination Of No Significant Hazards For Administrative Changes (Group 4)*

*Criterion 1*

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change involves the surveillances for mitigating equipment. Therefore, it has no impact on probability. The proposed change also has no impact on the consequences of an accident because the criteria for operable RHR [residual heat removal] and SI [safety injection] pumps does not change. The change is only in the parameter that will be compared with the required criteria, the differential pressure instead of the discharge pressure.

*Criterion 2*

Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. Nothing is changed with regard to accident initiators. The surveillance criteria for the RHR and SI pumps, which are mitigating equipment, is unchanged. The proposed change can have no impact on accident initiators.

*Criterion 3*

Does the proposal involve a significant reduction in a margin of safety?

No. The proposal does not change the requirements for a pump to be operable. Only the parameter compared to acceptance criteria changes. The underlying criteria is unchanged. Therefore, there is no change in the margin of safety.

*Conclusion*

*It is concluded that operation of Cook Nuclear Plant units 1 and 2 with the changes proposed above does not involve any significant hazards as defined in 10 CFR 50.92*

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

*Local Public Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

*Attorney for licensee:* Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW, Washington, DC 20037

*NRC Project Director:* John N. Hannon

**Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan**

*Date of amendment requests:* June 15, 1995 (AEP:NRC:0896V)

*Description of amendment requests:* The proposed amendments would change the 18 month emergency diesel generator (EDG) surveillance test from a 24-hour run to an 8-hour run.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Per 10 CFR 50.92, a proposed change does not involve a significant hazards consideration if the change does not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated,

2. create the possibility of a new of different kind of accident from any accident previously evaluated, or

3. involve a significant reduction in a margin of safety.

*Criterion 1*

The safety function of the EDGs is to supply ac electrical power to plant safety systems whenever the preferred ac power supply is unavailable. Through surveillance requirements, the ability of the EDGs to meet their load and timing requirements is tested and the quality of the fuel and the availability of the fuel supply are monitored. Reduction of the 24 hour run to 8 hours will not reduce the surveillance factors under consideration and will sufficiently exercise the EDG and its support systems to identify potential conditions that could lead to performance degradation. Based on these considerations, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

*Criterion 2*

The proposed changes do not involve physical changes to the plant or changes in plant operating configuration. The changes only involve the reduction of 18 month 24 hour EDG surveillance test duration. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

*Criterion 3*

Although the duration of the 18 month 24 hour EDG surveillance test would be reduced, the EDG components will continue to be sufficiently exercised such that the ability to detect incipient and degraded conditions will be maintained. The proposed changes have been determined to be compatible with our plant operating experience and commensurate with past surveillance test results. Based on these considerations, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

*Local Public Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

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*NRC Project Director:* John N. Hannon

**Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska**

*Date of amendment request:* June 15, 1995

*Description of amendment request:* The proposed amendment would revise the definition for logic system functional test and revise the surveillance interval for emergency core cooling system logic system functional testing from 6 months to 18 months.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed revisions to change the Cooper Nuclear Station (CNS) Emergency Core Cooling System (ECCS) logic system functional testing surveillance intervals from once/6 months to once/18 months do not involve a significant increase in the probability or consequences of an accident previously evaluated. The change in surveillance interval to once/18 months is necessary to coincide with scheduled refueling outages. The expansion of the scope of the logic system functional tests will ensure that once/18 months all contacts providing an automatic safety function in the ECCS logic systems will be tested. Revising the test frequency to once/18 months will prevent CNS from being required to install jumpers and/or test blocks during power operation, temporarily rendering various safety functions inoperable, and potentially challenging safety systems.

This proposed change will not result in any hardware changes to the facility, nor will it introduce any new mode of operation. Conversely, not changing the surveillance frequency would contribute to a slight, but measurable increase in the probability of an accident. Therefore, this change will not result in a significant increase in the probability of any accident previously evaluated.

This change will not result in a significant increase in the consequences of any accident previously evaluated. The District has evaluated the change in logic system reliability due to the increased proposed surveillance interval and determined it to be negligible. This conclusion is supported by a review of the surveillance history associated with the ECCS logic system functional tests which demonstrates that the logic systems perform reliably. Therefore, this change will not result in a significant reduction in the reliability or performance of the ECCS, and therefore, will not result in a significant increase in the consequences of any accident previously evaluated.

The change to the definition for ''Logic System Functional Test'' will not result in an increase in the probability or consequences of any accident previously evaluated. This change will only provide clarification of the definition for performing these tests.

These changes are also consistent with the NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4,'' dated September, 1992. Therefore, these changes have been previously reviewed and accepted by the NRC, and have been implemented at other plants.

2. Does the proposed change create the possibility for a new or different kind of accident from any accident previously evaluated?

The proposed changes revise the ECCS logic system functional testing surveillance intervals and the definition of that testing to be consistent with the Standard Technical Specifications, and therefore reflect current NRC guidance. The proposed changes do not involve any plant design changes nor any new mode of operation. Therefore, these proposed changes do not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change create a significant reduction in the margin of safety?

The proposed changes to the CNS ECCS logic system functional testing surveillance intervals do not create a significant reduction in the margin of safety. As discussed above, the District has revised its logic system functional testing to ensure that all contacts providing an automatic safety function in the ECCS logic systems are tested during this surveillance; thus, this change in testing scope will ensure that all essential functions in these logic systems are periodically tested.

The proposed changes will extend the ECCS logic system functional testing intervals to coincide with refueling outages. This will prevent CNS from being required to install jumpers and/or test blocks during power operation which would temporarily defeat safety system capability, and have the potential of challenging plant safety systems and/or degrading logic system reliability. The District has also determined that the change in test frequency will have a negligible impact on logic system reliability. Therefore, since these changes will continue to ensure the reliability of the ECCS logic systems, and thereby the capability of those systems to respond to accidents, these proposed changes do not create a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Auburn Public Library, 118 15th Street, Auburn, NE 68305

*Attorney for licensee:* Mr. G.D. Watson, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68602-0499

*NRC Project Director:* William D. Beckner

**Northeast Nuclear Energy Company (NNECO), Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut**

*Date of amendment request:* June 15, 1995

*Description of amendment request:* The proposed amendment would change the definition for an alteration of the reactor core to one that is consistent with the intent of the improved standard technical specifications. The proposed amendment also makes administrative changes to several technical specification pages.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

Revising the definition of core alteration would not affect the probability or consequences of a fuel handling accident, since the movement of fuel within the reactor vessel would still be considered a CORE ALTERATION. Additionally, movement of a fuel assembly continues to be performed under the supervision of a senior licensed operator. Therefore, the potential for inadvertent positioning of a fuel assembly would not be affected by the change to the definition of a core alteration.

Other activities which were not specifically excluded as core alterations in the existing technical specifications are now excluded. These activities do not affect the reactivity of the core.

Based upon the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

All required systems will continue to operate as before. Therefore, there is no possibility of a new or different kind of accident. The change in definition of a core alteration cannot create the possibility of a new type of accident since those activities which affect reactivity and could affect the initiating events for accidents will remain classified as core alterations.

3. Involve a significant reduction in the margin of safety.

Refueling operations which have the potential to alter the reactivity potential of the core will continue to be defined as core alterations. The margin of safety associated with those evolutions will not be altered as a result of the revised definition. As a result of the revised definition, evolutions which take place within the reactor vessel core region with the vessel head installed, or with the reactor vessel completely defueled, will not be considered core alterations. This does not constitute a reduction in the margin of safety since there is no impact on core reactivity potential during these conditions.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

*Attorney for licensee:* Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270

*NRC Project Director:* Phillip F. McKee

**North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire**

*Date of amendment request:* June 7, 1995

*Description of amendment request:* The proposed amendment would increase the temperature limit below which reactor coolant sampling and analysis for dissolved oxygen is not required. Specifically, the temperature limit stated in the footnotes to Technical Specification Surveillance Requirement 4.4.7 and to Table 3.4-2 would be increased to 250°F from 180°F.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed changes merely increase the temperature limit below which sampling of reactor coolant for dissolved oxygen and maintaining the dissolved oxygen below the specified limit would not be required. The proposed limit is consistent with data which shows that there is no significant oxygen-induced corrosion to reactor coolant system (RCS) components at or below the limit. The changes do not affect the manner by which the facility is operated and do not change any structures, systems, or components. Since there is no change to the facility or to the way it is operated, there is no effect upon the probability or consequences of any accident previously analyzed.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the manner by which the facility is operated or change any structure, system, or component. The proposed changes merely raise the temperature limit above which dissolved oxygen must be maintained within the specified limit. The changes are consistent with data for oxygen-induced corrosion of RCS components.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes are consistent with data for oxygen-induced corrosion of RCS components.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

*Attorney for licensee:* Thomas Dignan, Esquire, Ropes & Gray, One International Place, Boston MA 02110-2624

*NRC Project Director:* Phillip F. McKee

**Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California**

*Date of amendment requests:* June 29, 1995 (Reference LAR 95-04)

*Description of amendment requests:* The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, to add Mode 1 applicability to TS 3/4.4.2.2, ''Safety Valves - Operating,'' and to change the low-temperature overpressure protection (LTOP) system enable temperature for Mode 4 applicability from 323 degrees F to 270 degrees F in TS 3/4.4.2.1, ''Safety Valves - Shutdown.''

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no effect on plant operation. The proposed changes correct the applicability of TS 3/4.4.2.2, consistent with the NRC safety evaluation for License Amendments (LAs) 98 for Unit 1 and 97 for Unit 2, and LAs 100 for Unit 1 and 99 for Unit 2 dated March 9, 1995, and April 13, 1995, respectively.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature. Further, the proposed changes would not result in any physical alteration to any plant system, and would not be a change in the method by which any safety-related system performs its function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed administrative changes correct TS 3/4.4.2.2 applicability, consistent with previous NRC review and approval of LAs 98 and 97 and LAs 100 and 99, as described in the associated safety evaluations. Further, these proposed changes have no effect on current operating methodologies or actions that govern plant performance.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

*Local Public Document Room location:* California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

*Attorney for licensee:* Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120

*NRC Project Director:* William H. Bateman

**Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania**

*Date of amendment request:* June 5, 1995

*Description of amendment request:* The proposed changes will revise Technical Specification (TS) Section 3/4.1.5, ''Standby Liquid Control System,'' (SLCS), to remove the minimum flow rate requirement for the SLCS pumps from TS Section 3/4.1.5.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS change will remove the minimum flow rate requirement for the Standby Liquid Control System (SLCS) pumps from Technical Specifications Section 3/4.1.5. The proposed TS change does not involve any physical change in the plant configuration or the SLCS pumps operation. The SLCS is not used during normal plant operation; therefore, there is no impact on any accident initiators. The proposed TS change does not change the plant response to transients in any way that could increase the likelihood of an accident. The consequences of previously evaluated accidents are not affected since the SLCS pumps and the balance of the SLCS will continue to perform as designed, in accordance with the Anticipated Transient Without Scram (ATWS) Rule specified in 10CFR50.62. The SLCS pumps will continue to be tested periodically for operability in accordance with TS 4.0.5 Surveillance Requirements for American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B & PV) Code Class 2 pumps, and the testing frequency remains unchanged.

Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS change will remove the minimum flow rate requirement for the Standby Liquid Control System (SLCS) pumps from Technical Specifications Section 3/4.1.5. The SLCS and the SLCS pumps will continue to function as currently designed. There are no physical changes being performed to the SLCS or plant configuration. The proposed TS change does not introduce a new failure mode for the SLCS pumps. Physical and electrical redundancy and separation criteria are not impacted by this proposed TS change. There is no change to the Redundant Reactivity Control System (RRCS) logic which could create an accident or transient of a different type.

Therefore, the proposed TS change does not create the possibility of a new or different kind of accident, from any accident previously evaluated.

3. The proposed TS change does not involve a significant reduction in a margin of safety.

The following TS Bases were reviewed for potential reduction in the margin of safety:

3/4.1.5 Standby Liquid Control System

4.0.5 Surveillance Requirements

The margin of safety as defined in the TS Bases will remain the same. The specific flow rate requirement for the Standby Liquid Control System (SLCS) pumps is being removed from the TS since the Anticipated Transient Without Scram (ATWS) equation ensures acceptable flow rates. The SLCS pumps, which are safety-related, are not physically modified or impacted by the proposed TS change. The pumps will continue to be tested for operability, in accordance with TS 4.0.5 Surveillance Requirements for ASME B & PV Code Class 2 pumps, and the testing frequency remains unchanged. This testing will ensure that the SLCS pumps operate in accordance with the existing design basis for the SLCS.

Therefore, the proposed TS change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464

*Attorney for licensee:* J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

*NRC Project Director:* John F. Stolz

**Sacramento Municipal Utility District (SMUD), Docket No. 50-312, Rancho Seco Nuclear Station, Sacramento County, California**

*Date of amendment request:* June 20, 1995

*Description of amendment request:* The proposed amendment (PA-190) would permit SMUD to change the reviewer qualifications of the Permanently Defueled Technical Specification (PDTS) D6.5.3 from those required by ANSI N18.1-1971, Section 4.4 to those of Section 4. In addition, PDTS D6.9.6b, Environmental Reports, would be changed to permit annual reporting instead of the current semi-annual schedule.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has reviewed the proposed changes against each of the no significant hazards consideration criteria in 10 CFR 50.92, and, based on their safety analysis, concludes:

A significant increase in the probability or consequences of an accident previously evaluated will not be created, because the proposed PDTS changes (1) are administrative in nature, (2) have no effect on any credible accidents previously evaluated in the Rancho Seco Defueled Safety Analysis Report (DSAR) (i.e., the dropped fuel assembly accident, the loss of off-site power condition, or a radwaste tank rupture), (3) will not reduce the effectiveness of the reviews conducted because the Rancho Seco Qualified Reviewer training program ensures Qualified Reviewers have adequate skills to competently perform the required reviews and the Plant Review Committee will continue to conduct their second level review function, and (4) will only affect the timing and management of the required Environmental Reports submittals to the NRC.

PA-190 will not create the possibility of a new or different type of accident than previously evaluated, because the proposed PDTS changes (1) do not modify the configuration of the facility or affect facility operation during the PDM [permanently defueled mode], (2) are administrative in nature, and (3) do not provide any new mechanisms by which an accident can occur.

The proposed PDTS amendment will not involve a significant reduction in the margin of safety, because the proposed changes do not affect the operation of Rancho Seco or any plant systems. Also, The PDTS bases do not rely on (1) Qualified Reviewer qualification requirements or (2) submittal of PDTS D6.9.6b Environmental Reports to the NRC to provide a margin of safety for plant operation during the PDM. The Rancho Seco Qualified Reviewer program relies on training and not the ANSI N18.1 qualification requirements to ensure the PDTS D6.5.3 required reviews are competently performed. Therefore, the proposed changes will not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis, and based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration. Local Public Document Room location: Central Library, Government Documents, 828 I Street, Sacramento, CA 95814

*Attorney for licensee:* Dana Appling, Esq., Sacramento Municipal Utility District, P. O. Box 15830, Sacramento, CA 95852-1830

*NRC Project Director:* Seymour H. Weiss

**South Carolina Electric & Gas Company, South Carolina Public ServiceAuthority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina**

*Date of amendment request:* June 19, 1995

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TS) to delete the scheduler requirements for Type A testing (Overall Integrated Containment Leakage Rate) to be performed at 40 plus or minus 10 month intervals and to delete the schedular requirements for Type B and C tests to be performed at 24 month intervals.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

There is no increase in the probability of an accident since there is no work planned that would affect containment integrity. The testing of containment isolation valves and other containment penetration sealing devices is not postulated as an accident precursor or initiating event.

Type A testing is capable of determining the total leakage from both local leak paths as well as gross containment leakage paths. Our Type B and C testing has consistently provided accurate leakage rates for valves and penetrations.

Administrative controls govern maintenance and testing such that there is very low probability that unacceptable maintenance or alignments can occur. After maintenance on containment isolation valves (CIVs) and penetrations, a local leak rate test (LLRT) is required to be performed. All work on valves also requires that an independent valve lineup be performed. As a result, Type A testing is not required to accurately quantify the leakage through containment penetrations.

Any specific exemptions to the requirements of Appendix J will require approval by the NRC before implementation. The proposed change in itself does not affect reactor operations and does not change radiological consequences.

Therefore, this proposed change does not involve a significant increase in the possibility or consequences of an accident previously evaluated.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The proposed TS change request (TSCR) does not involve any physical changes to the plant, affect the operation of the plant, or change testing methods or acceptance criteria. The history of containment testing verifies that containment integrity has been maintained.

The scheduler change that is proposed should not significantly decrease the level of confidence in the ability of the reactor building to limit offsite doses to allowable values. No accident or malfunction can be the result of the change in test schedule or frequency.

Since the proposed TSCR will not directly impact equipment, procedures or operations, the changes will not create the possibility of any new or different kind of accident from any previously evaluated.

3. The margin of safety has not been significantly reduced.

The reason for performing ILRTs [integrated leakage rate tests] is to assure that the leakage paths are identified, and any accident release will be restricted to those paths assumed in

the safety analysis. The purpose for the schedule is to assure that containment integrity is verified on a periodic basis.

Revising the schedule does not mean that containment integrity will be compromised. Type B and C testing will still be performed. The requirements in 10 CFR 50 Appendix J still require the testing to be performed periodically.

The testing previously performed has shown that acceptable results were obtained. The ILRT results minus the LLRT results demonstrate that most of the increases in leakage are the result of LLRT increases. These changes in Type B and C leakage are tracked and corrective action is initiated at a specific action level.

Therefore, the margin of safety has not been significantly reduced.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

*Attorney for licensee:* Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

*NRC Project Director:* Frederick J. Hebdon

**South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina**

*Date of amendment request:* June 19, 1995

*Description of amendment request:* The proposed amendment would revise the Technical Specifications to change the required test frequency for the reactor building spray nozzle flow test from once per five years to once per ten years.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

This change does not effect the probability or consequences of an accident. The Reactor Building Spray System is normally idle, with the exception of testing. The possibility of the introduction of foreign material or corrosion products to restrict flow is minimized because of the use of 304 stainless steel as construction material. This change results in an extension of the testing periodicity only.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

This change results in an extension of the testing periodicity only and does not result in an accident not previously evaluated.

3. The margin of safety has not been significantly reduced.

The Reactor Building Spray System is normally idle, with the exception of testing. The possibility of the introduction of foreign material or corrosion products to restrict flow is minimized because of the use of 304 stainless steel as construction material. Industry wide spray system reliability, as demonstrated by the performance of these tests, justifies this change in the frequency of the nozzle flow test. This results in an extension of the testing periodicity only and will not significantly reduce a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

*Attorney for licensee:* Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

*NRC Project Director:* Frederick J. Hebdon

**Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee**

*Date of amendment request:* June 29, 1995 (TS 95-14)

*Description of amendment request:* The proposed change would, under certain stated administrative controls, allow both sets of containment personnel airlock doors to be open during core alterations and fuel movements. The administrative controls that would be added to Limiting Condition for Operation 3.9.4.b would allow both airlock doors to be open if one personnel airlock door in each airlock is capable of closure, and one train of the Auxiliary Building Gas Treatment System is operable in accordance with Specification 3.9.12. In addition, proposed changes to Surveillance Requirement 4.9.4 and 4.9.4.a would replace the requirement to determine that the containment building penetrations are in the ''closed/isolated'' condition with the need to determine that they are in the ''required'' condition, and delete the requirement to verify that the penetrations are in their required condition and the requirement to test the Containment Ventilation isolation valves ''within 100 hours prior to the start of'' core alterations or movement of irradiated fuel in the containment building. Related changes to the Bases would supply amplifying information.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 3.9.4, Containment Building Penetrations, would allow the containment personnel airlocks (PALs) to be open during fuel movement and core alteration. The PALs are not an initiator to any accident. The position of the PAL doors (open or closed) during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

All doses from a fuel handling accident (FHA) for the proposed change remain well below the 10 CFR 100 limits. The proposed change will reduce the dose to workers inside containment in the event of a FHA by allowing more rapid egress from containment. The wear on the PAL doors will significantly be decreased; therefore, increasing the reliability of the PAL doors in the event of an accident.

Since the probability of a FHA is not affected by the airlock door positions, and the doses remain within acceptable limits, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

As previously stated, the PAL doors are not accident initiators. The open PAL doors do not represent a significant change in the configuration of the plant; therefore, does not create a new or different type of accident from any previously analyzed.

3. Involve a significant reduction in a margin of safety.

The margin of safety provided for an FHA inside containment remains well below the 10 CFR 100 limits. Therefore, this proposed change to allow the PAL doors to remain open during fuel movement or core alterations does not involve a significant reduction in the margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

*Attorney for licensee:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

*NRC Project Director:* Frederick J. Hebdon

**Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington**

*Date of amendment request:* September 2, 1992

*Description of amendment request:* The proposed amendment revises the surveillance criteria for the source range monitors (SRMs) to incorporate a more conservative signal-to-noise (S/N) ratio, as recommended by General Electric for this system.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

1.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The SRM instrumentation is not assumed to be an initiator of any analyzed event. The SRM instrumentation provides monitoring of neutron flux levels to give the control room operator early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality. As such, action could be taken on the indication to avert or minimize the consequences of the event. However, the SRM function is not relied upon in any design bases or transient analysis. Rod motion interlocks and other instrumentation are relied on in the accident analysis to avert an accident. The change in acceptable count rate and signal-to-noise ratio preserves the confidence level of the General Electric design. As a result, the consequences of any analyzed events are unaffected because the change does not alter any system or component design assumptions or operation. Therefore, no significant increase in the probability or consequences of an accident previously evaluated will be involved.

2.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change in SRM count rate and S/N ratio values does not change modes of plant operation or require physical modifications. The WNP-2 design basis accident and transient analyses do not rely on the SRMs to assume plant safety. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

3.

The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety. The design basis to assure SRM operability is based on an instrument count rate that will assure the SRMs will provide early indication of subcritical multiplication with a 95-percent confidence level. Requiring the count rate to be greater than or equal to 0.7 counts per second (cps) with a S/N ratio greater than or equal to 20, or greater than or equal to 3 cps with a S/N ratio greater than or equal to 2 (vs. a count rate of greater than or equal to 0.5 cps with a S/N ratio greater than or equal to 2 in current TS) ensures the design 95-percent confidence level is maintained when verifying SRM operability. Therefore, the margin of safety is not affected by this change.

Based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352

*Attorney for licensee:* M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

*NRC Project Director:* William H. Bateman

**Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington**

*Date of amendment request:* June 6, 1995

*Description of amendment request:* The proposed amendment would change Technical Specification 6.9.3.2. The change would add references to three topical reports describing analytical methods that may be used in determining reactor core operating limits for reload licensing applications.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not remove or modify existing Technical Specification requirements or safety limits. The Technical Specifications will continue to require operations within analyzed core operating limits and appropriate actions be taken when, or if, limits are exceeded. There will be no changes to the physical design of the plant as a result of adding the proposed references to Section 6.9.3.2. The results of analytical determination of core operating limitations is not assumed as the initiator of any analyzed event, and the approved safety analysis is still applicable. Therefore, the proposed amendment to Technical Specification 6.9.3.2 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not remove or modify existing Technical Specification requirements or safety limits. The Technical Specifications will continue to require operation within analyzed core operating limits and appropriate actions be taken when, or if, limits are exceeded. The technical methodology outlined in the three new reports is in accordance with the accepted principals, and the specific reports proposed for inclusion in the Technical Specifications by this request have been previously approved by NRC for use at WNP-2 as a basis for core reload analyses. Therefore, the proposed amendment to Technical Specification 6.9.3.2 does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

Plant safety limits are established through LCOs, limiting safety systems settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings and limits as a result of adding the proposed references to Section 6.9.3.2. The ability to mitigate the consequences of all accidents previously evaluated will be maintained and nuclear safety is not adversely affected because the technical methodology outlined in the three new reports is in accordance with accepted principals, and the specific reports proposed for inclusion in the Technical Specifications by this request have been previously approved by NRC for use at WNP-2 as a basis for core reload analyses. Therefore, the proposed amendment to Technical Specification 6.9.3.2 does not significantly reduce any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352

*Attorney for licensee:* M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

*NRC Project Director:* William H. Bateman

**Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington**

*Date of amendment request:* June 6, 1995

*Description of amendment request:* The proposed amendment would change the Index of the WNP-2 technical specifications by deleting reference to the Bases pages. Consistent with the requirements of 10 CFR 50.36(a), which states that the Bases shall not become part of the technical specifications, the Bases information will be consolidated into a controlled plant document.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative and do not remove or modify existing Technical Specification requirements or safety limits. There will be no changes to the physical design of the plant as a result of the proposed change. The Bases information, per 10 CFR 50.36(a), is not part of the Technical Specifications and will be consolidated into a controlled plant document. Future changes to the Bases will be evaluated per 10 CFR 50.59. Therefore, the proposed changes to the Technical Specification Index do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative and do not remove or modify existing Technical Specification requirements or safety limits. There will be no changes to the physical design of the plant or alteration of any operational practice as a result of the proposed change. The Bases information, per 10 CFR 50.36(a), is not part of the Technical Specifications and will be consolidated into a controlled document. Future changes to the Bases will be evaluated under 10 CFR 50.59. Therefore, the proposed changes to the Technical Specifications Index do not create the possibility of a new or different type of accident.

3. The proposed change does not involve a significant reduction in a margin of safety.

Plant safety limits are established through LCOs, limiting safety system settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings and limits as a result of modifying the Technical Specification Index. The ability to mitigate the consequences of all accidents previously evaluated will be maintained and nuclear safety is not impacted. The Bases information, per 10 CFR 50.36(a), is not part of the Technical Specifications and will be consolidated into a controlled document. Future changes to the Bases will be evaluated under 10 CFR 50.59. Therefore, the proposed amendment does not significantly reduce any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352

*Attorney for licensee:* M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

*NRC Project Director:* William H. Bateman

**Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington**

*Date of amendment request:* June 6, 1995

*Description of amendment request:* The proposed amendment would change Technical Specification (TS) 6.0, ''Administrative Controls'' for WNP-2. Specifically, the changes would (a) reflect Supply System titles for senior management throughout TS 6.0, (b) modify the Plant Operations Committee (POC) composition to specify members according to functional areas rather than by organizational titles (c) replace the Plant Manager as the POC Chairman with an individual appointed by the Plant General Manager, and (d) make an editorial correction.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a) the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The senior management title changes are title changes only and will not impact the plant safety responsibilities associated with these positions. The removal of the Plant Operations Committee (POC) organizational titles and replacement with functional areas, and the elimination of the Plant Manager as the POC Chairman, will not impact the POC function because membership qualifications will continue to be consistent with the unit staff qualifications in TS 6.3.1 for those POC members and alternates considered part of the unit staff. Those designated POC members and alternates not considered part of the unit staff will possess skills and knowledge commensurate with their organizational positions. The proposed change ensures that POC will continue to be comprised of personnel who are experienced, have varied expertise, and are involved in daily plant activities. In maintaining the qualification requirements for members of POC, the POC will continue to fulfill its review and advisory responsibilities specified in TS 6.5.1.6 and TS 6.5.1.7. The proposed changes do not involve any physical changes to plant systems, structures, or components (SSC) or the manner in which the SSC are operated, maintained, modified, tested, or inspected. The changes therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Because the proposed changes are of an organizational nature and their implementation does not involve physical changes to the plant SSC or the manner in which the SSC are operated and maintained, the proposed changes do not create the possibility of a new or different kind of accident. The proposed changes do not introduce any new modes of operation or alter system setpoints which could create a new or different kind of accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed change does not involve a significant reduction in a margin of safety.

The senior management title changes do not impact the management responsibilities or functions associated with ensuring plant safety. Changes proposed in the POC composition will allow the scope of available expertise to be expanded without changing the POC function or responsibilities. Maintaining the current level of personnel qualifications and experience ensures the POC will continue to meet its TS review and advisory requirements. The proposed changes will not impact the basis for any Technical Specification related to the establishment of, or maintenance of, nuclear safety margins. Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has revieywed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352

*Attorney for licensee:* M. H. Philips, Jr., Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

*NRC Project Director:* William H. Bateman

## Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

### Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

*Date of application for amendments:* March 28, 1994

*Brief description of amendments:* The amendments change the minimum condensate storage tank indicated level from 25 feet to 29.5 feet to ensure that the condensate storage tank contains a sufficient volume of water. In addition, an editorial change was made to Technical Specification 3.7.1.3 for Unit 3 to be consistent with Units' 1 and 2 technical specifications.

*Date of issuance:* July 6, 1995

*Effective date:* July 6, 1995

*Amendment Nos.:* Unit 1 - Amendment No. 94; Unit 2 - Amendment No. 82; Unit 3 - Amendment No. 65

*Facility Operating License Nos.* NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** June 8, 1994 (59 FR 29625) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 6, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

### Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

*Date of application for amendments:* April 6, 1995, as supplemented by letter dated June 7, 1995.

*Brief description of amendments:* These amendments involve improvements delineated in Generic Letter 93-07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans,'' changes in plant review board, and miscellaneous minor changes.

*Date of issuance:* July 7, 1995

*Effective date:* July 7, 1995

*Amendment Nos.:* Unit 1 - Amendment No. 95; Unit 2 - Amendment No. 83; Unit 3 - Amendment No. 66

*Facility Operating License Nos.* NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 23, 1995 (60 FR 27335) The June 7, 1995, letter provided clarifying information and did not change the initial no sigificant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 7, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

**Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois**

*Date of application for amendments:* February 16, 1993, as supplemented by letter dated May 2, 1995.

*Brief description of amendments:* This application upgrades the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specification contained in NUREG-0123, ''Standard Technical Specification General Electric Plants BWR/4.'' This application upgrades only Section 3/4.10 (Refueling Operations).

*Date of issuance:* June 23, 1995

*Effective date:* Immediately, to be implemented no later than December 31, 1995, for Dresden Station and June 30, 1996, for Quad Cities Station.

*Amendment Nos.:* 136, 130, 157, and 153

*Facility Operating License Nos.* DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 23, 1995 (60 FR 27337) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 23, 1995. No significant hazards consideration comments received: No Local Public

*Document Room location:* for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

**Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois**

*Date of application for amendments:* December 23, 1994

*Brief description of amendments:* The amendments revise the Technical Specifications by increasing the allowable U-235 enrichment of fuel to be stored in the new fuel storage vault.

*Date of issuance:* June 22, 1995

*Effective date:* June 22, 1995

*Amendment Nos.:* 164 and 152

*Facility Operating License Nos.* DPR-39 and DPR-48: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** February 15, 1995 (60 FR 8742) The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated June 8, 1995, and a Safety Evaluation dated June 22, 1995. No significant hazards consideration comments received: No

*Local Public Document Room location:* Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

**Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois**

*Date of application for amendments:* March 24, 1995

*Brief description of amendments:* The amendments recognize performing containment leakage rate tests in accordance with 10 CFR Part 50, Appendix J, and approved exemptions.

*Date of issuance:* June 30, 1995

*Effective date:* June 30, 1995

*Amendment Nos.:* 165 and 153

*Facility Operating License Nos.* DPR-39 and DPR-48: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 26, 1995 (60 FR 20516) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 30, 1995. No significant hazards consideration comments received: No

*Local Public Document Room location:* Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

**Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania**

Date of application for amendments: June 8, 1993, as supplemented June 15, 1995

*Brief description of amendments:* These amendments revise item 2 of Technical Specification 6.9.1.14, ''Core Operating Limits Report,'' for Unit 1 and Unit 2, to specify the use of the BASH methodology instead of an earlier Westinghouse methodology. The BASH methodology is a Westinghouse improved and updated methodology which can be used to evaluate a large break loss-of-coolant accident. The BASH methodology was approved by the NRC staff on November 13, 1986.

*Date of issuance:* June 27, 1995

*Effective date:* As of the date of issuance, to be implemented within 60 days

*Amendment Nos.:* 189 and 71

*Facility Operating License Nos.* DPR-66 and NPF-73: Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** July 7, 1993 (58 FR 36433) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 27, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

**Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana**

*Date of amendment request:* August 9, 1994

*Brief description of amendment:* The amendment changed the Appendix A Technical Specifications (TSs) by revising the Administrative Controls Section of the TSs for Waterford 3 by removing the functions under review and audit from the TSs and by relocating those items in the quality assurance program manual. In addition the amendment removed the review and audit functions for the emergency plan and implementing procedures, and security plan from the list of responsibilities of the plant operation review committee in the TSs. These requirements will be retained in emergency plan or security plan as appropriate.

*Date of issuance:* July 6, 1995

**Effective date:** July 6, 1995, to be implemented within 60 days.

*Amendment No.:* 109

*Facility Operating License No.* NPF-38. Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** September 14, 1994 (59 FR 47167) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 1995.No significant hazards consideration comments received: No. Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

**Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida**

*Date of application for amendments:* April 3, 1995

*Brief description of amendments:* These amendments will incorporate line-item TS improvements to Specifications 3/4.8.1 ''Electrical Power Systems-A.C. Sources,'' and 4.8.1.2.2 ''Electrical Power Systems-Shutdown.'' The changes are consistent with recommendations for Emergency Diesel Generator (EDG) Surveillance Requirements in NUREG-1366, and regulatory guidance provided in Generic Letter (GL) 93-05 and GL 94-01. This issuance also contains FPL's commitment to implement a maintenance program for monitoring and maintaining EDG performance for both St. Lucie Units consistent with 10 CFR 50.65 and the guidance of Regulatory Guide 1.160.

*Date of Issuance:* June 29, 1995

*Effective Date:* June 29, 1995

*Amendment Nos.:* 138 and 78

*Facility Operating License Nos.* DPR-67 and NPF-16: Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 10, 1995 (60 FR 24910) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003.

**Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia**

*Date of application for amendments:* November 19, 1995

*Brief description of amendments:* The amendments relocate the requirements of Technical Specification 3/4.3.4, Turbine Overspeed Protection, to Section 16.3 of the Vogtle Final Safety Analysis Report (FSAR). In addition, the surveillance intervals for exercising the high pressure turbine stop valves, the low pressure turbine intermediate stop valves and intercept valves, and the high pressure turbine control valves are extended after relocation to the FSAR.

*Date of issuance:* July 3, 1995

*Effective date:* As of the date of issuance to be implemented within 30 days.

*Amendment Nos.:* 88 and 66

*Facility Operating License Nos.* NPF-68 and NPF-81: Amendments revised the technical Specifications.

*Date of initial notice in* **Federal Register:** February 16, 1994 (59 FR 7689) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 3, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

**Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia**

*Date of application for amendments:* December 27, 1994

*Brief description of amendments:* The amendments revise the frequency of conducting leak testing of containment purge valves with seals made of resilient material from every 3 months to each refueling outage.

*Date of issuance:* July 7, 1995

*Effective date:* As of the date of issuance to be implemented within 30 days.

*Amendment Nos.:* 89 and 67

*Facility Operating License Nos.* NPF-68 and NPF-81: Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** February 1, 1995 (60 FR 6301) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 7, 1995. No significant hazards consideration comments received: No

*Local Public Document Room location:* Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

**Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas**

*Date of amendment request:* February 15, 1995.

*Brief description of amendments:* The amendments modified (by relocation to the Technical Requirements Manual) Technical Specification (TS) 3/4.3.3.7, Chemical Detection Systems, and TS 3/4.8.4.1, Electrical Equipment Protective Devices - Containment Penetration Conductor Overcurrent Protective Devices, and the associated Bases.

*Date of issuance:* July 6, 1995

*Effective date:* July 6, 1995, to be implemented within 30 days.

*Amendment Nos.:* Unit 1 - Amendment No. 76; Unit 2 - Amendment No. 65

*Facility Operating License Nos.* NPF-76 and NPF-80. The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** March 29, 1995 (60 FR 16189) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 6, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

**IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy, Center, Linn County, Iowa**

*Date of application for amendment:* March 28, 1995

*Brief description of amendment:* The amendment revises Technical Specifications (TS) Table 3.2-A by clarifying or correcting entries to the table. The amendment also revises the TS Bases to describe more clearly the logic arrangements in Table 3.2-A.

*Date of issuance:* June 14, 1995

*Effective date:* June 14, 1995

*Amendment No.:* 212

*Facility Operating License No.* DPR-49. Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 26, 1995 (60 FR 20519) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 14, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Cedar Rapids Public Library, 500 First Street, S. E., Cedar Rapids, Iowa 52401.

**Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan**

*Date of application for amendments:* March 31, 1995

*Brief description of amendments:* The amendments modify the Containment Ventilation System Technical Specifications (and associated Bases) to allow limited containment purge operation in Modes 1, 2, 3, and 4 for pressure control, ALARA [as low as is reasonably achievable], and respirable air quality considerations.

*Date of issuance:* June 23, 1995

*Effective date:* June 23, 1995

*Amendment Nos.:* 195 and 181

*Facility Operating License Nos.* DPR-58 and DPR-74. Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 26, 1995 (60 FR 20520) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 23, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

**Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook, Nuclear Plant, Unit No. 1, Berrien County, Michigan**

*Date of application for amendment:* March 17, 1995

*Brief description of amendment:* The amendment allows a one-time extension of the required test interval for the overall integrated containment leak rate test (Type A test). This extension allows the third Type A test of the second 10-year service period to be performed during the refueling outage that will follow the end of Cycle 15. Concurrently, the Commission has also granted a one-time schedular exemption to allow an extension of one cycle for the performance of the 10 CFR Part 50, Appendix J, Type A test.

*Date of issuance:* July 6, 1995

*Effective date:* July 6, 1995

*Amendment No.:* 196

*Facility Operating License No.* DPR-58. Amendment revises the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 26, 1995 (60 FR 20519) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 1995.No significant hazards consideration comments received: No. Local Public

*Document Room location:* Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

**Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota**

*Date of application for amendments:* December 5, 1994, as supplemented January 9, 1995 and May 15, 1995.

*Date of application for amendments:* The amendments revise the Prairie Island Technical Specifications to allow containment airlock doors to remain open during core alterations provided certain conditions are met. In its May 15, 1995, letter, the licensee withdrew the portion of its original application which dealt with containment penetrations during core alterations. The staff granted the licensee's request to withdraw all aspects of its application concerning the opening of containment penetrations during core alterations.

*Date of issuance:* July 3, 1995

*Effective date:* July 3, 1995

*Amendment Nos.:* 119/112

*Facility Operating License Nos.* DPR-42 and DPR-60. Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** February 1, 1995 (60 FR 6306). The January 9 and May 15, 1995, letters provided updated Technical Specification pages and clarifying information in response to discussions with the staff during various teleconferences conducted during the review process. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 3, 1995. No Significant hazards consideration comments received: No.

*Local Public Document Room location:* Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

**Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska**

*Date of amendment request:* February 10, 1995

*Brief description of amendment:* The amendment relocates the requirements for the incore instrumentation (ICI) system from the technical specifications to the Updated Safety Analysis Report (USAR).

*Date of issuance:* June 26, 1995

*Effective date:* June 26, 1995

*Amendment No.:* 167

*Facility Operating License No.* DPR-40. The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** March 15, 1995 (60 FR 14025) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

**Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California**

*Date of application for amendments:* April 19, 1995 (LAR 95-03)

*Brief description of amendments:* The amendments would allow an emergency diesel generator (EDG) hot restart test within 5 minutes of a 2-hour run at the continuous rating instead of an EDG loss of offsite power load sequencing test within 5 minutes of the 24-hour endurance run.

*Date of issuance:* June 26, 1995

*Effective date:* June 26, 1995

*Amendment Nos.:* Unit 1 - Amendment No. 105; Unit 2 - Amendment No. 104

*Facility Operating License Nos.* DPR-80 and DPR-82: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 23, 1995 (60 FR 27340) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 26, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

**Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California**

*Date of application for amendments:* February 16, 1994, as supplemented by letter dated April 25, 1995 (Reference LAR 94-05)

*Brief description of amendments:* The amendment revises Technical Specifications 3/4.7.2, ''Steam Generator Pressure/Temperature Limitation,'' 3/4.7.7, ''Snubbers,'' 3/4.7.8, ''Sealed Source Contamination,'' 3/4.7.11, ''Area Temperature Monitoring,'' and 3/4.7.13, ''Flood Protection,'' in accordance with the Commission's final policy statement for relocation of current technical specifications to licensee controlled documents that do not satisfy any of the policy statement criteria.

*Date of issuance:* July 6, 1995

*Effective date:* July 6, 1995

*Amendment Nos.:* Unit 1 - Amendment No. 106; Unit 2 - Amendment No. 105

*Facility Operating License Nos.* NPF-10 and NPF-15: The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 13, 1994 (59 FR 17603) The April 25, 1995, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 6, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

**Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York**

*Date of application for amendment:* February 22, 1995

*Brief description of amendment:* The amendment modifies operability and surveillance requirements for the reactor vessel overfill protection instrumentation that initiates feedwater pump turbine and main turbine trips on high reactor vessel water level. The NRC staff has determined that the proposed Technical Specification (TS) changes will have no adverse impact on plant safety and will enhance the current TSs by adding operability requirements for the reactor vessel overfill protection system. Therefore, the proposed TS changes are acceptable.

*Date of issuance:* June 19, 1995

*Effective date:* As of the date of issuance to be implemented within 30 days.

*Amendment No.:* 225

*Facility Operating License No.* DPR-59: Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 10, 1995 (60 FR 24915) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 19, 1995.No significant hazards consideration comments received: No Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

**Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey**

*Date of application for amendment:* July 28, 1994 and December 15, 1994

*Brief description of amendment:* This amendment makes changes to TS Section 3/4.8.1 ''AC SOURCES.'' The staff found it appropriate to combine these two applications into one amendment. The amendment removes the surveillance requirements, methodology and frequency for Emergency Diesel Generator (EDG) fuel oil from the TS and relocates them in a controlled plant procedure, VSH.SS-CA.ZZ-0013(Q) ''Procedures for Testing Diesel Fuel and ι2 Fuel Oil at Artificial Island for PSE&G Nuclear Operations.'' The changes also delete an unnecessary lab test for the fuel oil and extend the surveillance frequency from once per 92 days to once per 184 days. In addition and in accordance with 10 CFR 50.90, this amendment removes TS Surveillance Requirement 4.8.1.1.2.h.1 in order that PSE&G can utilize plant-controlled programs to govern diesel generator maintenance. To ensure procedural consistency and reduce the impact of this change on Hope Creek procedures, the remaining Surveillance Requirements of TS 4.8.1.1.2.h are not renumbered.

*Date of issuance:* June 29, 1995

*Effective date:* June 29, 1995

*Amendment No.:* 74

*Facility Operating License No.* NPF-57: This amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** August 31, 1994 (59 FR 45034) and April 26, 1995 (60 FR 20526) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

**Public Service Electric & Gas Company, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey Date of application for amendment: April 4, 1995**

*Brief description of amendment:* The amendment allows a one-time interval extension for the Type A test required by 10 CFR Part 50, Appendix J. Instead of conducting the test during the twelfth refueling outage, it can now be conducted during the thirteenth refueling outage, but no later than June 1997.

*Date of issuance:* July 5, 1995

*Effective date:* July 5, 1995

*Amendment No.:* 171

*Facility Operating License No.* DPR-70: This amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** May 23, 1995 (60 FR 27341) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 5, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

**Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee**

*Date of application for amendments:* April 6, 1995; supplemented May 26, 1995 (TS 94-19)

*Brief description of amendments:* The amendments revise action statements to eliminate starting of emergency diesel generators in order to verify their operability whenever one of the required electrical power sources is inoperable or a diesel is inoperable unless the diesel inoperability is due to a common cause failure.

*Date of issuance:* June 29, 1995

*Effective date:* June 29, 1995

*Amendment Nos.:* 205 and 195

*Facility Operating License Nos.* DPR-77 and DPR-79: Amendments revise the technical specifications.

*Date of initial notice in* **Federal Register:** April 26, 1995 (60 FR 20529) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1995.No significant hazards consideration comments received: None

*Local Public Document Room location:* Chattanooga-Hamilton County Library,1101 Broad Street, Chattanooga, Tennessee 37402.

**The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio Date of application for amendment: September 27, 1993 and December 16, 1994**

*Brief description of amendment:* The amendment revised the Technical Specification Section 6.8.1, ''Unit Staff Qualifications,'' to make it consistent with the current requirements of Part 55 of Title 10 of the Code of Federal Regulations.

*Date of issuance:* June 27, 1995

*Effective date:* June 27, 1995

*Amendment No.:* 70

*Facility Operating License No.* NPF-58: This amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** December 8, 1993 (58 FR 64604) and February 1, 1995 (60 FR 6310). The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated February 28, 1995, and a Safety Evaluation, dated June 27, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

**Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri**

*Date of application for amendment:* July 16, 1993

*Brief description of amendment:* The amendment revises the Technical Specifications (TS) 3/4.8.1.1 and 3/

4.8.1.2. The changes address the minimum required storage volumes of the Emergency Fuel Oil storage and day tanks.

*Date of issuance:* July 6, 1995Effective date: July 6, 1995

*Amendment No.:* 100

*Facility Operating License No.* NPF-30: Amendment revises the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 13, 1994 (59 FR 17607) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 1995. No significant hazards consideration comments received: No.

*Local Public Document Room location:* Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

**Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.**

*Date of application for amendments:* January 24, 1995

*Brief description of amendments:* The amendments change the ''as-found'' test criterion for the pressurizer safety valves from plus or minus 1% to plus or minus 3%

*Date of issuance:* June 29, 1995

*Effective date:* June 29, 1995

*Amendment Nos.:* 200 and 200

*Facility Operating License Nos.* DPR-32 and DPR-37: Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 12, 1995 (60 FR 18631) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1995.No significant hazards consideration comments received: No

*Local Public Document Room location:* Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

**Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington**

*Date of application for amendment:* July 12, 1994

*Brief description of amendment:* The amendment modifies the technical specifications (TS) to remove instrument response time limit tables for the reactor protection system, isolation actuation system, and emergency core cooling system from the TS. The affected instrument response time limit tables will be located in the Final Safety Analysis Report (FSAR).

*Date of issuance:* June 26, 1995

*Effective date:* June 26, 1995, to be implemented within 30 days of issuance.

*Amendment No.:* 139

*Facility Operating License No.* NPF-21: The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** August 31, 1994 (59 FR 45036). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

**Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin**

*Date of application for amendments:* January 24, 1995, as supplemented by letters dated February 24, April 25, May 24, and June 1, 1995

*Brief description of amendments:* These amendments revise Point Beach Nuclear Plant Technical Specification (TS) Section 15.6.5, ''Review and Audit,'' and TS Section 15.7.8, ''Administrative Controls.'' The quality assurance audit frequencies and the section on emergency plan reviews are relocated to other documents, and the period for radioactive effluent reporting is increased to annual. In addition, the references to ''Semiannual Monitoring Report'' are changed to ''Annual Monitoring Report'' throughout TS Sections 15.7 and 16.5. Administrative changes are also included.

*Date of issuance:* July 5, 1995

*Effective date:* July 5, 1995

*Amendment Nos.:* Unit 1 - 162: Unit 2 - 166

*Facility Operating License Nos.* DPR-24 and DPR-27. Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** March 1, 1995 (60 FR 11142). The February 24, April 25, May 24, and June 1, 1995, submittals provided supplemental information that did not change the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 5, 1995.No significant hazards consideration comments received: No.

*Local Public Document Room location:* Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

**Notice Of Issuance Of Amendment To Facility Operating License And Final No Significant Hazards Consideration Determination**

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

**Commonwealth Edison Company, Docket No. 50-295, Zion Nuclear Power Station Unit 1, Lake County, Illinois**

*Date of amendment request:* May 17, 1995

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TS) to allow 154 steam generator tubes that potentially exceed the repair or plugging criteria to remain in service for the remainder of the current Unit 1 operating cycle.

*Date of publication of individual notice in* **Federal Register:** May 25, 1995 (60 FR 27798)

*Expiration date of individual notice:* June 26, 1995

*Local Public Document Room location:* Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

**Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois**

*Date of amendment request:* June 14, 1995

*Description of amendment request:* The proposed amendments would revise the Technical Specifications (TS) to allow the hot restart sequence loading test of the emergency diesel generators to be performed independent of the 24 hour endurance test.

*Date of publication of individual notice in* **Federal Register:** June 30, 1995 (60 FR 34308)

*Expiration date of individual notice:* July 31, 1995

*Local Public Document Room location:* Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Dated at Rockville, Maryland, this 12th day of July 1995.

For the Nuclear Regulatory Commission

**Jack W. Roe,**

*Director, Division of Reactor Projects - III/IV,Office of Nuclear Reactor Regulation*

[Doc. 95-17565 Filed 7-18-95; 8:45 am]

**BILLING CODE 7590-01-F**

## Availability of Draft Application Format and Content Guidance and Review Plan and Acceptance Criteria for Non-Power Reactors

The U.S. Nuclear Regulatory Commission (NRC) is in the process of developing for Non-Power Reactors (NPRs) a ''Format and Content for Applications for the Licensing of Non-Power Reactors'' (F&C) and a ''Standard Review Plan and Acceptance Criteria for Applications for the Licensing of Non-Power Reactors'' (SRP). The NRC has made available a draft of Chapter 16, ''Other License Considerations,'' of the F&C and SRP documents for comment. Other draft chapters will be made available for comment as they are completed.

Copies of these chapters have been placed in the NRC Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555. Single copies of these documents may be requested in writing from Alexander Adams, Jr., Senior Project Manager, US Nuclear Regulatory Commission, MS: 0–11–B–20, Washington, DC 20555. Comments on this chapter should be sent by October 12, 1995 to the Director, Non-Power Reactors and Decommissioning Project Directorate at the above address.

Dated at Rockville, MD, this July 12, 1995.

For the Nuclear Regulatory Commission.

**Seymour H. Weiss,**

*Director, Non-Power Reactors and Decommissioning Project Directorate, Division of Project Support, Office of Nuclear Reactor Regulation.*

[FR Doc. 95–17721 Filed 7–18–95; 8:45 am]

**BILLING CODE 7590–01–M**

**[Docket No. 50–400]**

## Carolina Power & Light Co.; Notice of Withdrawal of Application for Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Carolina Power & Light Company (the licensee) to withdraw its August 20, 1993, application for proposed amendment to Facility Operating License No. NPF–63 for the Shearon Harris Nuclear Power Plant, Unit No. 1, located in New Hill, North Carolina 27562.

The proposed amendment would have revised the TS to allow the repair of degraded steam generator tubes by sleeving as an alternative to removing the tube from service by plugging. The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the **Federal Register** on September 15, 1993 (58 FR 48378). However, by letter dated July 5, 1995, the licensee withdrew the proposed change.

For further details with respect to this action, see the application for amendment dated August 20, 1993, and the licensee's letter dated July 5, 1995, which withdrew the application for license amendment. The above documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Dated at Rockville, Maryland, this 12th day of July 1995.

For the Nuclear Regulatory Commission.

**Ngoc B. Le,**

*Project Manager, Project Directorate II–1, Division of Reactor Projects—I/II, Office of Nucler Reactor Regulation.*

[FR Doc. 95–17719 Filed 7–18–95; 8:45 am]

**BILLING CODE 7590–01–M**

**[Docket Nos. 50–424 and 50–425]**

## Georgia Power Company, et al.; Vogtle Electric Generating Plant, Units 1 and 2; Exemption

### I

Georgia Power Company, et al. (the licensee) is the holder of Facility Operating License Nos. NPF–68 and NPF–81 for the Vogtle Electric Generating Plant, Units 1 and 2. The licenses provide, among other things, that the licensee is subject to all rules, regulations, and orders of the Commission in effect now and hereafter.

The facility consists of two pressurized water reactors near Waynesboro in Burke Country, Georgia.

### II

Title 10 of the Code of Federal Regulations (10 CFR) 73.55, ''Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage,'' paragraph (a), in part, states that ''The licensee shall establish and maintain an onsite physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.''

Section 73.55(d), ''Access Requirements,'' paragraph (1), specifies that ''The licensee shall control all points of personnel and vehicle access into a protected area.'' Section 73.55(d)(5) requires that ''A numbered picture badge identification system shall be used for all individuals who are authorized access to protected areas without escort.'' Section 73.55(d)(5) also states that an individual not employed by the licensee (i.e., contractors) may be authorized access to protected areas without escort provided the individual ''receives a picture badge upon entrance into the protected area which must be returned upon exist from the protected area * * *''

The licensee has proposed to implement an alternative unescorted access control system that would eliminate the need to issue and retrieve badges at each entrance/exit location and would allow all individuals with unescorted access to keep their badges when departing the site.

An exemption from 10 CFR 73.55(d)(5) is required to allow contractors who have unescorted access to take their badges offsite instead of returning them when exiting the site. By letter dated February 14, 1995, the licensee requested an exemption from the requirements of 10 CFR 73.55(d)(5) for this purpose.

### III

Pursuant to 10 CFR 73.5, ''Specific exemptions,'' the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions in this part as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest. Pursuant to 10 CFR 73.55, the Commission may authorize a licensee to provide alternative measures for